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Abstract

DEVELOPMENT OF CERAMIC NUCLEAR MATERIALS IN THE CZECHOSLOVAK SOCIALIST REPUBLIC

Dušan Jakeš

The development of ceramic nuclear materials in the Czechoslovak Socialist Republic started with a concept of the DOR reactor. Uranium dioxide powder was studied as a raw material for the preparation of dense ceramic fuel. Besides, attention was paid to the densification by means of standard powder-metallurgy techniques and to the relations between physico-dispersion properties of the powder and its behaviour during fabrication.

All basic stages of UO_2 powder preparation were studied in detail beginning with ADU precipitation, drying, heat treatment to give UO_3 or U_3O_8, and also reduction to UO_2. A correlation between the powdery product properties and a number of factors affecting the preparation was found. The relations between aggregate structure, specific surface area, and chemical reactivity found out during these research works enable easy control of a technological preparation process and correction of inconvenient properties.

UO_2 powder samples with specific surface areas ranging from about 2.5 to about 13 sq.m/g were compacted by cold-pressing. It was found that the compactability of UO_2 powder can be described mathematically. On the basis of coherence between the fine structure and the compactability of the powder, the density after pressing or the pressure needed to reach a definite density can be estimated. Sintering of the compacts was studied especially in reduction atmosphere of hydrogen in the temperature range from 1200° to 1800° and its sintering properties were also found for this process. On the basis of physico-chemical measurements resulting density value accurate within 1% can be estimated and optimum pressing and sintering conditions determined at the same time.

Several theoretical questions concerning ADU structure changes during precipitation and the post-reduction non-stoichiometric oxygen location in UO_2 structure were also investigated.

Preparation methods of non-pyrophoric uranium powder with well controllable grain size were worked out for the synthesis of other types of ceramic fuels. A process of UO_2-SiO_2 co-reduction with calcium hydride was investigated. Grain growth of the metal phase during heat exposure was measured in a U ÷ UC cermet system.

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Abstract

RESEARCH PROGRAMME AND OPERATING EXPERIENCE ON WWR-S REACTOR

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J. Urbanec, F. Voženílek and V. Zajíc

The report deals with the reactor use by scientific laboratories and with the works aimed at its improvement and development.

(a) The first part of the report, which is devoted to the reactor exploitation, contains a survey of all experimental works carried out in the fields of nuclear physics, reactor physics and nuclear technology.

These investigations refer first of all to the study of radiation capture of thermal neutrons using the method of gamma ray spectrometry of different nuclei, determinations of the spin characteristics of the levels in compound nuclei by angular correlation measurements in radiation capture, the studies on obtaining polarized neutron beams by reflection on a cobalt mirror and the design of a neutron spectrometer with a mechanical chopper. In the field of nuclear technology the studies on two experimental loops in the reactor core are given. The first loop is aimed at the study of a carbon dioxide cooled fuel element and the second at the corrosion study of some materials in water situated in a radiation field.

(b) In the second part studies are reported on reactor physical characteristics as well as on the changes and improvements which were or will be performed in the reactor. The measurement results of absolute fluxes of thermal and resonance neutrons using activation methods and the distribution of fast neutron absolute fluxes and fast neutron spectra using threshold detectors in irradiation channels of the reactor core are given. For increasing experimentation and irradiation capacity the project of reactor core reconstruction will be realized which will permit doubling the nominal value of the reactor power. The project of a facility for automatic sample inserting and removing in or from the reactor core, the potentiality of using

an apparatus for measuring corrosion losses of stainless steel and aluminium reactor components using non-destructive methods are described and the experiences are surveyed which were gained during the operation of the instrumentation for a complex automatic reactor control using moving detectors.

(c) In the conclusion the experiences are summarized comprising the co-operation of the reactor laboratory with other scientific laboratories, experimental capacity of the WWR-S reactor as well as experimental work programming.

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Abstract

METHODICS OF THE OPTIMISATION OF PARAMETERS OF A NUCLEAR POWER STATION WITH A HEAVY WATER MODERATED GAS-COOLED REACTOR

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and Karel Januška

1. Introduction

The submitted methodics of the optimisation of parameters for a nuclear power station is founded upon a consistent applying of the only one criterion of optimality for the using up of all degrees of freedom for to achieve the optimum of this criterion.

Because of its intricacy the optimisation so considered requests urgently the use of a digital computer and because of it, there is requested the creation of a full mathematic model.

2. A general formulation of the task

The mathematic model is composed of three parts, each of a different character.

The first of them is the sole criterion of optimality, which is a synthetic economic indicator created in harmony with the level and the needs of national economy. It may be expressed as a function of a certain number of physical-technical and economic quantities. It is universal for all optimisational computations of similar kind.

The second component is cost relation, expressing the dependence of the cost items upon the physical-technical parameters. The relations are dependent upon the level of the industry, and upon the material base and may be deduced on the ground of experiences with the building of nuclear power stations of the given type.

The third component of the mathematic model is creating relations, which describe mutual dependencies of individual, physical and technical

parameters of the equipment of the nuclear power station of the given conception variant.

From the mathematic point of view the model is a system of n equations (complemented with technical limitations), with m variables, and so with m-n degrees of freedom. The task commits in the finding of such values of independent variables, where the criterion reaches the optimal value.

3. Application of methodics

There were selected as a criterion of optimality the computational expenses that respect not only the simple but also the enlarged reproduction. The cost relations are deduced on the ground of the technical design of the first Czechoslovak nuclear power station with a heavy-water-moderated gas-cooled reactor. Physical-technical relations were elaborated for several variants of the same type, and differ what concerns the diameter of the vessel of the reactor, the arrangement of the active zone, and the type of the secondary circuit. Optimality calculations including the finding out of optima were made on a digital computer NE 803.

4. Conclusions

The results we have achieved made possible to find out an optimal variant of a nuclear power station and the basic influence of the initial parameters and so the impulses for further improvements of the nuclear power station of the type mentioned above were acquired.

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Abstract

The SEFOR Reactor
- Aspects of International Collaboration -

by

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The Southwestern Experimental Fast Oxide Reactor, a sodium-cooled, plutonium-fueled experimental fast oxide reactor with a thermal power of 20 MW, is under construction at Fayette-Ville, Arkansas, USA. It is being built by the General Electric Atomic Power Equipment Department for the Southwest Energy Associates, a considerable contribution to the financing being made by the Gesellschaft für Kernforschung m.b.H., Karlsruhe[3]. The purpose of this reactor is the demonstration of the reliable operating characteristics and inherent safety features of a large, plutonium-uranium-fueled fast reactor. A large research and development programme will be performed in this facility. Sponsored by the Atomic Energy Commission, it will be carried out by General Electric under subcontract to Southwest Atomic Energy Associates.

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[1] Kernforschungszentrum Karlsruhe, Germany

[2] Southwest Atomic Energy Associates of Little Rock, Arkansas, USA

[3] The Karlsruhe fast breeder development project is carried out in the frame-work of an association with Euratom.

The main incentives for this extensive cooperative effort were the following:

The Karlsruhe Center has an active programme in the field of the fast power breeder development and had already in 1960 envisaged the construction of a fast dynamical test reactor.

Southwest Atomic Energy Associates, a group of 17 utilities serving the South of the USA, has sponsored nuclear energy development for a long time. In view of the growing interest in fast power breeder reactors, they have decided to support the development of this reactor type.

The General Electric Atomic Power Equipment Department - like the Karlsruhe Center - had realized the vital importance of a programme and carried out a design study which was first presented at the 1961 IAEA Conference at Vienna.

In the years 1961 to 1963 a series of negotiations between the above partners, the US AEC and the Euratom Commission were carried out which finally led to an agreement covering the construction of the reactor and the performance of the research and development programme.

In view of the different interests of the participating groups and of the fact that SEFOR is the first reactor to be built within the USA with substantial support from a non-US group, the negotiating partners were faced with difficult problems and many new solutions had to be found.

After a detailed discussion of the above mentioned technical and scientific incentives and a review of the historical development of the project, the paper will describe the legal relations among the various partners. It further covers the problems of financing and the organisation which was created in order to guarantee to all partners the necessary influence and control. Finally the question of the use of the obtained information and patents and the exchange of personnel and knowledge will be discussed, with special consideration to existing international agreements.

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Abstract

Research Reactors in the Federal Republic of Germany

by

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presented by K. Wirtz

Kernforschungszentrum Karlsruhe

A survey paper on the utilisation of research reactors in the Federal Republic of Germany is given. The following classes of research reactors are covered:

1. Major reactors for applied and basic research and for isotope production at the three main nuclear energy research centers Karlsruhe, Jülich and Geesthacht
2. Reactors for basic research and training at München and Frankfurt universities and at the Hahn-Meitner institute in Berlin
3. Small reactors for training purposes at various universities
4. Argonaut-type reactors for reactor physics studies at several reactor development laboratories

1) Kernforschungszentrum Karlsruhe
2) Institut für Kernphysik der Universität Frankfurt/M.
3) Reaktorstation Geesthacht-Tesperhude
4) Laboratorium für Technische Physik der T.H. München
5) Kernforschungsanlage Jülich
6) Hahn-Meitner-Institut für Kernforschung Berlin

The various low-energy critical and subcritical assemblies in use for reactor physics and engineering studies are not included in the survey.

The paper includes a short technical description of each reactor (in tabular form) and some additional information on major experimental facilities, such as loops, neutron spectrometers and other devices of special interest. Operational problems relevant to the utilisation of the reactors are briefly discussed. Applications are described in the following fields

- Basic research (neutron and nuclear physics, solid-state physics, radiation chemistry, radio-chemistry and radio-biology)
- Applied research and engineering (reactor physics, shielding, fuel element and material testing)
- Isotope production
- Training

A substantial part of the research programmes is carried out in collaboration between various institutions. Universities and other scientific organisations cooperate with the major nuclear energy centers in the fields of basic research and training, whilst some of the engineering and applied science research projects are operated jointly by industry and the major reactor centers. Several programmes are carried out within the framework of association contracts with Euratom. The more important of these cooperational efforts will be briefly reviewed.

Among the future developments to be discussed are the MZFR project now under construction at Karlsruhe - a multipurpose reactor for power production and fuel element testing - and several smaller special purpose research reactors to be built at various laboratories. No further large research and test reactors are presently envisaged for the next years. However, there will be a considerable effort to improve further the utilisation of the existing reactors and - by close international cooperation - to utilize irradiation capacities available in foreign research and test reactors. There is also a considerable interest in participation in the ENEA Very High Flux Reactor project.

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Abstract

Development of the Superheat Reactor in the Federal Republic of Germany

by

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Allgemeine Elektricitäts-Gesellschaft

The development of the boiling water reactor to a reactor with nuclear superheating of steam represents one of the most interesting improvements of today's power reactors in technical and economical respect. A number of various concepts has been proposed for this purpose. Since 1960 AEG in the Federal Republic of Germany is working on the development of an integrated single-zone superheat reactor with combined boiling and superheating elements. The decisive advantages of this concept are represented by excellent shut-down and emergency cooling characteristics, elimination of additional structure material in the core, uniform core design, fixed correlation of superheat and boiling power independent from control rod positioning and nearly independent from power, low maximum fuel temperatures, low reactivity flooding coefficient and relatively low steam content on the boiling side, favorable for the future development to higher outputs per liter. Hereby a relatively complicated steam rooting and the occurrence of considerably different thermal expansions in the element have to be overcome.

Since 1960 the developmental work has resulted in the design of a prototype power station with 100 MWth. The power station is to be erected in the direct vicinity of the boiling water reactor Kahl. The following problems of the development program have been treated:

Development of Fuel Elements - Theoretical Work

Theoretical study of element behaviour, design of models and com-

pilation of computer codes.

Fabrication of Fuel Elements

Experimental fabrication of superheat fuel elements, development work with the aim to master vibration and swaging techniques for cladding materials, concentric vibration of cladding tubes, making of weld joints with different superheat cladding materials and different types of weld seams.

Testing of Fuel Elements

Testing of combined boiling and superheat elements under full operation-temperatures of the cladding and under the full steam pressure, thermocycling tests with the fuel elements, creep tests with cladding materials under full temperature conditions, creep tests with irradiated cladding materials for superheat elements at full operating temperature during irradiation as well as during the creep testing, operation of the first combined boiling-superheat elements in the Kahl Experimental Loop since December 10, 1963.

Corrosion Tests

Corrosion studies in autoclaves under temperatures of up to 700°C, influence of oxygen and chlorine content on stress corrosion cracking, experiments in test solutions, results of experiments with special melts particularly for use in superheat reactors.

Reactor Physics Work

Elaboration of calculating methods and computer codes for superheat reactor lattices, evaluation of superheat reactor lattices in the AEG Test Reactor in particular in view of the flooding reactivity coefficient, measuring and calculation of the thermal utilization factor in the large Seitz-Wigner-Cells of the superheat reactor.

Thermodynamic Work

Theoretical studies, elaboration of computer codes, measuring of burnout, steam slip, steam distribution and pressure drop of steam-water mixtures in the test rig under full pressure and temperature conditions, theoretical and experimental study of steam-water separation.

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Abstract

Development Project of a Pebble Bed Power Reactor

by
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By an association between EURATOM, the Kernforschungsanlage Jülich of North-Rhine Westphalia, and BROWN BOVERI/KRUPP Reaktorbau GmbH the line of pebble bed reactors initiated by the latter company is being extended towards a high-temperature reactor with Thorium-Uranium-Cycle for big power stations. The project financed by these partners and the Federal Ministry of Scientific Research of the Federal Republic of Germany is intended to investigate the fundamental problems of the application of this reactor type to large scale power production and to lead to the complete design for a 500 MW thermal prototype-reactor.

With this project the basic philosophy will be revised in the light of technical and economic feasibility. The technical progress achieved with the general development of fuel elements for gascooled high-temperature reactors has reduced the principle difficulties arising from the contamination of the primary circuit and, therefore, it seems possible to have the full benefit of high temperature and high burn-up along with good neutron economy. Major efforts of research and development will be directed tc fuel elements, irradiation and corrosion behaviour of graphite, mass transfer problems within the primary circuit, flow characteristics of a large pebble bed core, purification of the primary circuit, burn-up measurements of single balls, and neutron physics of Uranium-Thorium-Carbon-Systems. The reactor vessel, the heat exchangers and blowers, the fuel handling system, and the arrangement of these components will be studied in regard to technical performance, possible extrapolation to larger systems, and economics.

The programme provides for capsule irradiation tests of various fuel element types and reactor materials, such as graphite and steel, in the test reactors BR 2 at Mol and FRJ 2 at Jülich. The AVR-reactor at present under construction at Jülich is to be operated as a test facility to support the research and development work as well as the design effort. This will be particularly the case for fuel elements both from the operation charge and test charges resulting from further development.

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Abstract

A Sodium Cooled Power Reactor Experiment Employing Zirconium Hydride Moderator

by

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The characteristics and some details of the design for a 20 MWe sodium cooled reactor experiment are to be described. The heterogeneous core will be fuelled by steel-canned, slightly enriched UO_2, using zirconium hydride as the moderator. The heat transfer system has been designed for a maximum sodium temperature of 560°C to generate steam of 510°C at 85 atm pressure. Essentially all system parts in contact with high temperature sodium are to be constructed of a low alloy ferritic steel that does not decarburize on hot sodium.

The design has already been supported by an extensive experimental development effort over a period of several years. The program comprises a number of reactor-physics experiments (neutron spectra measurements, pulsed subcritical and critical experiments), the development and the investigation of zirconium hydride moderator elements, the testing and the evaluation of a low-alloy ferritic material for high-temperature sodium applications, systems- and core components development by means of various sodium loops including a 5-thermal-Megawatt Sodium-Test-Facility (permitting the operation of sodium heated steam generators up to 540°C and 160 atm pressure), investigation of steam tube ruptures in sodium heated steam generators etc. Some results of the experimental program are to be given.

Finally a comprehensive survey of the economical and technical aspects of the reactor type considered will be outlined.

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Abstract

The MOSEL Reactor Concept
- A Relative Evaluation -

by

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The MOlten Salt Epithermal reactor concept (MOSEL) utilizes uranium and thorium fluorides dissolved in molten fluoride salts, and concerns a two-fluid, two-region reactor operating in the intermediate-to-fast neutron energy range. The concept involves a unique application of the molten salt technology developed in the USA during the past decade. Reactor operation is at low pressure (10 - 15 atmospheres) and high temperature (outlet coolant temperature of 650° C); the fluid nature of the core and blanket salts facilitates fuel movement and processing. The core fluid consists of UF_4 dissolved in NaF plus BeF_2 and is contained in tubes or plates of a high-nickel-containing alloy. Fuel fluid is circulated for purposes of mixing, fuel addition and removal, and to improve the effective thermal conductivity of the fuel salt. The core is internally cooled, with blanket fluid used as the core coolant. The blanket salt contains ThF_4 dissolved in NaF plus BeF_2. The core coolant transfers energy to the steam generator through an intermediate molten salt circuit. Fuel processing consists of uranium recovery by fuel fluorination, and is applied to both the blanket and core fluids. In the blanket cycle, uranium-free salt is returned to the blanket region, while the recovered UF_6 is reduced to UF_4 and fed to the core region. Core processing consists of uranium recovery and discard of fission-product-containing salt. The combination of blanket-salt recycle, low core-fuel volume, and processing limited to uranium recovery makes the fluoride volatility process

+ On assignment from Oak Ridge National Laboratory, Oak Ridge/ Tenn.

uniquely applicable, and permits economic integration of the processing and reactor facilities leading to low unit processing costs (about $ 5/kg Th equivalent).

The fluid nature of the fuel, the simple processing scheme, the construction of the core without the presence of radioactive fuel, and the use of blanket fluid as the coolant permit reactor power to be increased without significant change in nuclear performance. This is done by extending the core in one direction, perpendicular to coolant flow, which also permits modular-type construction. The primary areas where further information is needed concern the energy variation of eta for U^{233}; the scattering and absorption cross sections of component salts and structural materials; the corrosion behaviour of fluoride salts as a function of temperature, salt composition and structural material; the effective thermal conductivity of various salts as a function of operating conditions; and the design of steam generators utilizing molten salts (salt melting temperature about 500° C). Nonetheless, based on present information, the MOSEL reactor concept can have economic breeding ratios in the range of 1.0 - 1.1, fuel cycle costs less than 1.5 mills/kwhe and power costs of 6 mills/kwhe in power plants of 500 Mwe rating, utilizing on-site processing and fuel recycle, and employing capital charges of 15 and 10 %/year for depreciating and non-depreciating items, respectively.

Relative to other reactor concepts having low power costs when very large nuclear power stations are considered, MOSEL systems appear to be economically competitive. At the same time, the MOSEL concepts appears superior for combining on-site fuel recycle, breeding ratios above unity, and low power costs in relatively small nuclear plants.

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Abstract

The Karlsruhe Fast Breeder Project *

by

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and

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The basic approach of a fast breeder reactor design of the second generation using non metallic fuel elements is outlined. The lower density and the trend to lower Pu contents result in lower power densities. This influences the choice of a proper coolant. The use of non metallic fuel elements brings an emphasis of the lower tail of the neutron spectrum. Most of the power coefficients, in particular the Doppler coefficient, is strongly influenced by that fact and this means that the basic problem of dynamic stability must be considered. The high burn-ups required for the second generation of fast breeder reactors are the basic problem for a proper fuel element design.

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* The Karlsruhe Fast Breeder Project is a common enterprise as an association of Gesellschaft für Kernforschung, Karlsruhe, and Euratom.

** The first part of the references is considered to be an integral part of this presentation. Members of the Karlsruhe team with more specific and published contributions are mentioned there. Apart from that all members of the Karlsruhe team are contributing heavily.

The desired closed fuel cycle brings the attention to the step of reprocessing and refabricating.

In the light of this reasoning the Karlsruhe Fast Breeder Project is explained. There are three main areas of activities:

a) Physics and Safety

Extended theoretical work is going on since the very beginning. The calculation of the Doppler coefficient and other power coefficients is one of the most important tasks. Apart from that a set of integral reactor experiments is prepared

SNEAK (Pu fueled fast critical facility)
STARK (coupled thermal fast reactor)
SUAK (fast pulsed subcritical assembly)
SEFOR (fast reactor for excursion tests)

b) Fuel Element Development

A fuel element for very high burn-up is being developed. The fuel itself and the cladding impose serious problems. Large-scale irradiation tests are prepared. In the same context the adjusted type of reprocessing and refabricating is considered and developed.

c) Choice of a proper coolant and the development of key hardware components

Besides Na, also He and dry steam are considered to be possible coolants. Conceptual design studies and the development of key components, like for example blowers, are carried out.

The more explicitly defined Fast Reactor Programme, to whom Euratom is associated, is complemented by some large-scale effort of the German industry. This complementary effort is shortly described.

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Abstract

The Reactor Installation of the Research Vessel of Kernenergie-Gesellschaft

by

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König, Arbeitsgemeinschaft Babcock-Interatom
I. Weisbrodt, Arbeitsgemeinschaft Babcock-Interatom

The research vessel of the "Gesellschaft zur Verwertung von Kernenergie in Schiffbau und Schiffahrt" (GKSS), Hamburg, was ordered at the Howaldtswerke, Kiel, in November 1962, and will be launched about April 1964. The reactor for this ship, ordered at the beginning of 1964 from the associated group of Babcock and Interatom, is an advanced type of a pressure water reactor, operating under saturation conditions at a reduced primary pressure of 63 ata, and being designed according to the integrated system.

The requirements for ship-reactors of any type are far more severe than for stationary installations. The available space is restricted and weight has to be saved as far as possible. In spite of this it is necessary to dimension the strength of all parts of the reactor and its supports for additional forces from acceleration. Under all movements of the ship at sea the operation must be reliable and the working of all safety devices without fail. As moreover a ship may be exposed to accidents like collision, grounding, capsizing, and sinking, it must be assured that even under such condition the reactor will not become a source of dangerous consequences.

The research vessel is over a certain length of the hull, the so-called reactor zone, reinforced by a special collision protection and a higher double-bottom. One compartment of this zone houses the containment built as a vertical cylindrical vessel which is supported from double-bottom and bulkheads in a statically determined way. While on the N.S. "Savannah" the containment is a horizontal cylinder requiring a greater length, the reactor compartment is here of square section. The secondary shielding which on the N.S. "Savannah" covered the containment is here applied on the surrounding bulkheads. Thus it is possible to inspect the whole containment from the outside. A further characteristic of the reactor, described later on, will permit even inspections on the inside.

Forward of the reactor compartment proper lies another compartment for supplementary installations and furthermore a shielded compartment for storage of burnt-up elements, i.e. an integrated service station.

The containment can be entered through an air lock and has several flooding flaps for pressure equalizing in case of sinking. Inside the containment the pressure vessel takes a central position within a shield water tank containing the primary shielding. The pressure vessel encompasses the complete primary cooling system including the heat exchangers. Only the three primary circulating pumps are on the outside being connected to nozzles. The integral construction, entirely different from the system used on the N.S. "Savannah" is a marked progress in several respects:

1. By the lack of outside fittings from primary coolant like pipe-lines, heat-exchangers and shut-off valves,

2. by improved safety against collision and radiation effects, and further

3. by the lower operating pressure of 63 ata as against 123 ata for the N.S. "Savannah".

There are some more advances in relation to the N.S. "Savannah" installation: All decontamination is performed inside the containment; the number of control rods is reduced from 21 to 12, the number of fuel elements from 32 to 16; the electric operating power is essentially lower; the primary circulating pumps are of the simple propeller type; the mounting and accessibility is substantially improved; there are three independent heat exchangers instead of two; safety and control devices are much simplified; an accident by uncontrolled access of cold water is impossible.
The primary circuit inside the pressure vessel passes through the core in the lower center position upwards within a chimney toward the surface of the water, and returns through the annulus between chimney and wall of the vessel. The upper part of this annulus contains the tube-bundle system of the heat exchanger. The control rods penetrate the cover of the pressure vessel and are operated from the outside. In addition to the three pump nozzles in the bottom there are three feedwater-inlet and three steam-outlet nozzles in the cylindrical portion of the pressure vessel. The heat exchanger is working in the once-through principle and delivers slightly superheated steam of 30 ata.

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Abstract

Krypton-85 in the Atmosphere

by

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According to measurements of several authors the atmospheric krypton-85 content increased by an order of magnitude since 1955. It has been stated previously**) that there is more krypton-85 in the atmosphere than one would expect from the estimated total energy release and the krypton-85 yield of all nuclear test explosions conducted so far. It seems probable therefore that atmospheric krypton-85 is primarily due to nuclear fuel treatment.

Since spring 1961 monthly krypton-85 samples have been collected in the Federal Republic of Germany***). These samples are taken directly from the oxygen-argon-xenon-outlet of an air liquefier. They are purified by fractional distillation. The xenon which is not removed from the samples is assayed by means of a mass-spectrometer, the krypton-85 activity is measured in a

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*) Gesellschaft für Linde's Eismaschinen

**) G. Delibrias and C. Jehanno, Bull. Inform. Sci.Techn., Paris, 30, 14 (1959);
O. Griesser und A. Sittkus, Z. Naturforschung 16a, 620 (1961).

***) D. Ehhalt et al., J. Geophys.Res. 68, 3817 (1963); and
D. Ehhalt Comptes Rendus du Colloque International sur la pollution radioactive des milieux gazeux, Saclay 13-16 Nov. 1963.

proportional counter. These closely spaced samples cover a period of strongly varying nuclear test activity.

Nevertheless, one finds a smooth increase in krypton-85 whose correlation with bomb testing is smaller than the experimental uncertainty. On the basis of the extremely high strontium-90 spring peak in 1963 it was possible to set an upper limit for the corresponding bomb contribution which should have shown up in the krypton-85 data if the bomb contribution were substantial. On the basis of available data on the Kr^{85}/Sr^{90} bomb yield and plausible assumptions on the worldwide dispersion of both nuclides, one would expect a flat hump of 5 - 10 % in the general krypton-85 increase which is indeed not outside the scatter of the measurements.

The krypton data presented therefore confirms strongly the suspicion that krypton-85 is primarily due to other sources, most probably to nuclear fuel treatment.

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Abstract

The Use of Power Reactors as Large-Scale Radiation-Sources for Radiation-Chemical Production

by

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In the core of power reactors, a high-intensity radiation is produced, most of which must be annihilated in the thermal and biological shield. Various possibilities of chemical utilization of this radiation have been shown. Such utilization may take place either in the "chemistry reactor", so called[1] (irradiation in the core) or in the "blanket reactor"[1] (irradiation in the reflector range). The use of nuclear reactors as large-scale radiation sources for radiation-chemical reactions requires a reactor giving a high γ-dose rate in a large volume (in order to achieve a satisfactory chemical production rate) and a neutron flux not exceeding a certain limit (to keep contamination by activation far below the MPC). These two requirements cannot yet be met within the scope of the suggestions made so far and with the reactors known at present. For this purpose, reactors of entirely novel design would have to be developed, which seems impossible in the near future.

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[1] Summary: KRAUCH, H.: Dechema Monographien, Vol. 42, p. 145 (1962).

We are suggesting in this paper the "compound reactor" as a new conception. It is characterized by the fact that the chemical reaction takes place outside the reactor vessel; in this case, the chemical plant takes over, in whole or in part, the function of the thermal shield. Such "compound reactor" permits economical chemical production since it utilizes only the "waste"-radiation of the reactor which has otherwise to be annihilated in the radiation shield.

Any technically perfected power reactor may be used, without the necessity of interfering in the neutron economy. The utilization of fuel, therefore, is not reduced. A reactor of this type should have low γ-self-absorption of the core, of the reflector and, if necessary, of the thermal shield, and should be operated at a low moderator pressure. Boiling water reactors and pressurized water reactors are not suitable as "compound reactor" types because they require too large amounts of material for the pressure vessel; reactors according to the GCR principle are also not suitable because of their poor power density and large amounts of moderator. A suitable reactor type, however, would seem to be, for instance, a pressure-tube reactor with small amounts of moderator and gas cooling at medium pressures, since in this case the γ-absorption in the core and in the container material can be made as small as possible, and the neutron absorption as high as possible.

The proposed "compound reactor" has the following advantages:

1) Energy generation and radiation-chemical production are practically independent of each other and take place under optimum conditions.
2) Reactor safety is not impaired.
3) The irradiation space for radiation chemistry is not restricted.
4) The neutron flux can be shielded to a given limiting value without interfering in the neutron economy.
5) Free selection of the reaction parameters (pressure, temperature) and the material of the reaction vessels for the radiation-chemical reaction.

6) Facilitated maintenance of the chemical plant.

For some technically interesting reactions which take place in the liquid phase, the amounts that can be produced are calculated for several pressure-tube reactors (CVTR, AKB). E.g., using the radiation data given in the literature, the following production amounts can be estimated for the AKB reactor (300 MW_{th}) in a reaction volume of approximately 30 m^3: hexadecane sulphonic acid 4,000 tons per month; ethyl bromide 70,000 tons per month.

The compound reactor as a combination of energy generation and chemical production may well contribute to economic utilization of nuclear energy.

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Abstract

THE DEVELOPMENT OF BERYLLIA-BASED FUELS FOR H.T.G.C. REACTOR SYSTEMS

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As part of a High Temperature Gas Cooled Reactor feasibility study by the Australian Atomic Energy Commission (Alder and Roberts - this Conference), a materials research programme has been in progress since 1958 to evaluate the potential of beryllium oxide as a fuel diluent in dispersion fuels and also as a moderator material. The programme has been designed around an all-ceramic core, cooled by carbon dioxide, employing highly rated, fission-product-retentive, high burn-up fuels with a maximum fuel surface temperature of 1000°C. Although various fuel cycles are being considered, experimental work on materials has been confined mainly to a U^{235}/Th cycle and dispersion fuels consisting of homogeneous fuel particles of $(U\ Th)O_2$ in a matrix of BeO. This work has involved fuels with from 3-30 vol % of irregular or spherical fuel particles ranging in size from 10-200 microns.

The paper surveys the Australian work directed towards understanding and evaluating the behaviour of beryllia and beryllia-based fuels in relation to the conditions under which they might be used in the proposed reactor system. Attention is confined mainly to the problems of resistance to thermal stress, fission-product retention, corrosion resistance, irradiation behaviour, chemical reprocessing and overall fuel cycle costs and their relation to fabrication variables, microstructure, temperature and neutron environment.

Irradiation experiments in essentially thermal fluxes have shown that stability under fission-product damage is excellent up to 2×10^{21} fissions/cm^3 of fuel particle at temperatures from 600-850°C with thermal stresses up to 30 000 psi.

A more serious problem is fast-neutron damage to the beryllia matrix which causes powdering at high doses (Elston and Caillat, 2nd UN Geneva Conference, Paper 15/P/1159). At temperatures up to 800°C this has been

shown to be due to displaced atoms causing anisotropic growth of crystals, high intergranular stresses and eventually microcracking. Microcracking in the fuel matrix must be avoided since it causes loss of strength, reduction in thermal conductivity and increased release of fission-products. The onset of microcracking is shown to be dependent on material structure, in particular the grain-size, as well as the irradiation temperature, dose and dose-rate. On the basis of the results, analytical expressions have been developed to allow predictions of the property changes to be expected in beryllia under a variety of reactor conditions. Fine grained beryllia (grain size less than 5 microns) should withstand a dose of 2×10^{21} nvt (fast) at 400°C and greater than 10^{22} nvt (fast) at 800°C and above; this should allow beryllia to be used satisfactorily as a fuel diluent in dispersion fuels, but not as fixed moderator to last the reactor life.

Fabrication techniques for obtaining high-density fine-grained bodies with adequate resistance to thermal stress and satisfactory fission-product retention are reported together with possible techniques for chemical re-processing of the fuel. Corrosion losses at temperatures up to 1000°C due to water vapour in the coolant are shown to be almost negligible.

The preliminary work has given confidence in the technical performance of all-ceramic beryllia-based fuel provided it is used in the form of simple shapes which facilitate fabrication and handling problems and which suit the intrinsically brittle nature of the material. A progress report is given of further work directed towards a more complete engineering evaluation of spherical-shaped fuel elements for a pebble-bed reactor concept.

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Abstract

METALLURGICAL ASPECTS OF ELECTROSLAG WELDING OF REACTOR VESSEL STEELS

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Thesis of a paper for the Third International United Nations Conference on the Peaceful Uses of Atomic Energy.

1. One of the main problems connected with the construction of nuclear power plant equipment is the production of reactor pressure vessels, their main parameters and geometric size being very important for their production.

In order to secure high quality plates for these pressure vessels, the necessary materials are made of relatively small ingots; this is why the single parts of the pressure vessel are made of more components.

Considering the plate thickness, it is most economical to make the single vessel components using electroslag welding.

2. The resulting mechanical as well as structural properties of electroslag welded joints are conditioned not only by the base metal type, but also by three main metallurgical factors characterized as follows:

(a) by the way of physico-chemical processes between the air medium and the slag pool as well as between the slag pool and the liquid weld metal;

(b) by the final primary weld metal structure;

(c) by the susceptibility of the base metal to grain coarsening in the heat affected zone of the welded joint.

3. The said metallurgical factors along with the base metal type characterize also the susceptibility of the welded joint to nitrogen aging which was confirmed by mechanical tests on welds after various heat treatment and detailed structural analysis.

4. To eliminate the influence of aging on the mechanical properties of welded joints and particularly on the transition temperature of the characteristic weld zones, it is necessary on the basis of well known structural and chemical changes in the base material and weld metal to choose especially suitable steels and filler materials for electroslag welding (welding flux and wire).

5. Good results were achieved using Mo alloyed and Al and Ti desoxidized base metal, this being welded with a specially developed flux and welding wire with a small quantity of V added.

6. The stability of electroslag welded joint from the standpoint of nitrogen aging was confirmed by aging tests and structural analysis of the welded joint. Recommendations for the choice of base metals, filler materials, suitable welding regimes and principal heat treatment of electroslag welded joints.

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Abstract

MODE OF OPERATION OF ATOMIC POWER STATION IN INTERCONNECTED SYSTEM AND THE ECONOMIC EFFECTS

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In atomic power stations according to the features of the reactor cooling two methods of the operation are possible. One of them, at present generally applied, when the parameters (pressure and temperature) of the thermal cycle in the function of electric rating are unchanged. With such a mode of operation the atomic power station is applicable as base load power station, or as a peak-load one. Thermodynamically justified is such a mode of operation when the parameters of the thermal cycle in the function of the rating are changed so that at peak load the parameters are lower and at part load they are higher. This mode of operation favourably uses the possibility that a reactor, lowering the temperature, is markedly overrateable. This way, the same power station generally can be operated with a good efficiency and at the time of peak demand, without heat storage, can be overrated. The paper deals with these two methods of operation, analyses the situation in an atomic power station, and the first costs of electric energy, taking into consideration the mutual influences of a new atomic power station, and the interconnected system. In the cost calculation the effects of the new power station on the rating and on the costs of the power stations of the interconnected system are considered. On the basis of economical investigation the two methods of operation are compared and conclusions are drawn, when the different methods of operation are economic. The investigation is extended to electric energy and heat-producing atomic power stations.

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Abstract

EFFECTS OF PLANT UTILIZATION ON THE ECONOMY OF HEAT-PRODUCING NUCLEAR PLANTS AND SUITABLE MEANS TO IMPROVE IT

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Present status of the economy of heat-producing nuclear plants in function of plant utilization. Comparative economic analysis of heat supply in back-pressure cycle and pure electricity generation in condensing cycle with the same small power reactor as heat source. Such an investigation is made necessary by the fact that many countries will - as the first step of their nuclear power programme - construct a small capacity reactor anyhow. As will be shown the heat-producing nuclear power plant is preferable in this case if plant utilization is high.

Evaluation of the various means to improve plant utilization:

Application of pass-out condensing turbines in heat-producing power stations. Calculation of limit values of plant utilization below which the pass-out condensing system is preferable to the pure back-pressure system. Demonstration of the increased potentialities of the pass-out condensing cycle in a nuclear plant as against conventional plant.

Combination of heat-producing nuclear power plant and fossil fuel fired peaking boiler plant. As will be shown, the application of a peaking boiler plant improves the economy of nuclear heat supply even while nuclear energy is not competitive with fossil fuels. Calculation of the optimal values of peaking boiler plant capacity in function of plant utilization. Effects of reactor capital cost reductions on optimum peaking plant capacity. Comparative analysis of conventional and nuclear heat-producing power plant showing that with the latter the application of a peaking boiler plant results in increased savings and the optimal value of peaking plant capacity is also increased.

Application of heat accumulators to meet the daily fluctuation of the heat demand. Description of the various systems of heat storage. Parametric calculation of the cost of heat sent out from nuclear versus conventional power plant using heat accumulators pointing to the increased importance of heat storage in nuclear plants.

Centralized heat supply to several heat consumers. Demonstration of the possibilities to improve plant utilization by supplying heat to several heat consumers from a central reactor plant. Possible effects of nuclear energy application on the siting of new heat consumers.

Other possibilities of improving the load factor of nuclear power plants, such as combination plants for space heating and air conditioning and the application of heat pump.

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Abstract

LIMITATION OF EXPOSURE TO WORKERS IN RADIATION EMERGENCIES

International Labour Office

Under radiation emergency conditions, as defined in this paper, there exists a requirement for prompt rescue or salvage action. Characteristically, such action may entail acceptance by workers of radiation doses much in excess of those acceptable under normal conditions. Action decisions in a particular emergency situation must be based upon an intelligent assessment of existing circumstances. The degree of urgency involved, rather than a rigid exposure limit applicable to all emergency situations, could be the important factor in operational decisions.

Acceptance by rescue workers of higher-than-normal radiation exposures, when not required by the degree of urgency encountered, is inherently undesirable. On the other hand, unreasoning fear or refusal on the part of such a worker to accept higher-than-normal exposures in the performance of essential rescue operations may be detrimental to his own safety, to that of his fellow workers, and to the public interest.

Under most circumstances, it is in the best interests of all concerned to restrict planned emergency exposures to less than 12 rems, as has been recommended by the ICRP. In general,

furthermore, necessary emergency actions should be designed to minimize higher-than-normal exposures, to the extent consistent with prompt and adequate response to the emergency situation. On the other hand, the urgency may be extreme; situations may arise in which lives will be lost if action is delayed. It is appropriate to consider whether, in such cases, doses higher than 12 rems should be accepted by rescue workers, particularly in view of the relatively small risk of significant biological injury at that level.

There is a present and continuing need for careful, sober deliberation and international debate on the subject of emergency exposure limitation. There is also a pressing need to provide adequate background material upon which the responsible individual can base a judgment on the relative risks associated with alternative courses of action in a radiation emergency.

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Abstract

THE POSSIBLE UTILISATION OF NUCLEAR REACTORS FOR SEA WATER DISTILLATION IN ISRAEL

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After about ten years of water resources development on a national scale, Israel is approaching the stage where about 90% of the total water available - 1,500 to 1,700 million m^3 per year - will be in use. (The present water demand is about 1,200 million m^3 per year). Also, part of the water available has a salt content exceeding the amount admissible for agriculture.

Continued development of Israel, a typical area of water scarcity, depends on new water sources, the biggest part of which could come only from desalination of sea water.

It is estimated that up to 200 million m^3 per year (about 600,000 m^3/day or 160 million G.P.D.) of converted sea water will be needed within the next ten years.

Three different types of desalination plants are in operation or being constructed in Israel;- using the distillation process, the freezing principle and electrodialysis. The planning of the sea-water conversion plants is directed by a Government-sponsored Sea Water Conversion Committee. A dual-purpose plant at Eilat, comprising a 6,700 kw back-pressure turbogenerator, the exhaust steam of which will be used to operate two 30-stage flash evaporator units, each producing 160 m^3/h (3,800 m^3 or 1 million gallons water per day), is scheduled to start operation by the end of 1964. Also at Eilat, the locally developed Zarhin freezing process is used in a pilot plant of four units with a total capacity of 40 m^3/h or 1,000 m^3/water per day (250,000 G.P.D.).

Several electrodialysis pilot plants for the desalination of brackish water

with a capacity of up to 20 m^3/h (125,000 G.P.D.) are also in operation or being built.

The most economical solution for sea water conversion seems to be to build future electricity generation units as dual-purpose plants, similar to, but of much higher capacity than, the Eilat plant, corresponding to the amount of extension required for existing power stations. The feasibility of nuclear steam generation for these extensions is under investigation.

The present total installed capacity of the power grid in Israel is 720 Mwe, the doubling period of the electricity demand being five to six years. The biggest generating units now under construction have a capacity of 125 Mwe.

The exhaust steam of a back pressure turbogenerator of 125 Mwe, if using the same high pressure steam conditions (140 atm. g., 530°C) as 125 Mwe condensing units, would be sufficient for heating up to four flash evaporator units, each of 22,500 m^3/day (6 million G.P.D.) capacity. These are the largest flash evaporator units available commercially to date. They will have a performance ratio (kg of distilled water per kg of heating steam) of approximately 12, with heating of the sea water (limited by scaling risk) to 110°C (230°F). The Eilat 30-stage flash evaporators will have a performance ratio of 8 and a maximum sea water temperature of 94°C (200°F).

If further developments in large flash evaporators make it possible to heat the sea water up to 135°C (275°F), or possibly even to 150°C (300°F), it may be economical to generate the evaporator-heating steam directly, without passing it through a back-pressure turbine.

Under an IAEA research contract, dual-purpose nuclear reactors for power and desalination purposes and process heat reactors for low pressure steam generation for desalination only will be compared technically and economically with conventional steam generation in Israel.

On the basis of different electricity credit and capital cost estimations, the costs per cubic metre of distilled water are calculated. Investment costs based on these alternatives and the water prices expected are given in tabular and diagrammatic form.

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REVIEW OF ENGINEERING WORK FOR THE DRAGON PROJECT

by

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The 20 MW(t) Dragon reactor experiment is a graphite moderated, helium cooled, high temperature reactor now in the last stages of construction at Winfrith, Dorset, England.

Criticality is expected during the second quarter of 1964 and construction work, carried out with the participation of a large number of contractors and organisations of the different signatory countries, started in April 1960.

This paper reviews the main aspects of the engineering work through the stages of design, manufacture and site construction and testing which have all been affected to different extents by the novel features peculiar to this type of reactor, such as the use of helium as a coolant, the use of an uncanned ceramic fuel, the high coolant temperatures and the extreme compactness of the plant.

One of the requirements affecting all the components of the primary circuit is the need to meet a very high standard of leak tightness, which would be normal on laboratory scale but which is novel for a large plant. Other requirements are the need to purify the helium coolant both from fission products and from chemical impurities, as the helium must be very pure to avoid chemical reactions with the hot graphite, and the need to measure the resulting extremely low levels of impurities during reactor operation.

The use of very pure helium has meant that special care has had to be taken with the selection of materials for bearings and gears because they are not operating in an oxidising atmosphere, as is usually the case.

In general, high standards of materials and workmanship have been specified to satisfy the safety and operational requirements and to ease the site construction problems of the reactor experiment which is, by its nature and purpose, more complex than a power reactor would need to be.

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Abstract

THE ITALIAN NUCLEAR POWER PLANTS

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In the first place, an outline is given on the main construction problems that arose during the installation of the three Italian nuclear power plants (SIMEA, SENN and SELNI) and a brief account is made of the acceptance tests concerning the nuclear and conventional components of the said plants.

The author then illustrates the problems which came up during the initial period of exploitation of the first two plants, in particular as regards the fuel cycle programme - whose optimization constitutes one of the principal factors of nuclear power stations - and the problems inherent to the transportation and chemical treatment of (the) irradiated fuel.

Still in connection with the three Italian nuclear plants, the author estimates the incidence which the capital cost (calculated on the basis of elements concurring to the assessment of direct and indirect costs), as well as the fuel, operation and maintenance costs have on the final cost of producible energy.

Lastly, the author describes the procedure used for the computation of the cost of nuclear kWh for such plants as are equipped with reactors on both natural and enriched uranium.